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AEP
INDIANA
MICHIGAN
POWER

February 17, 2004

AEP:NRC:2573-18
10 CFR 50.73

Docket No. 50-316

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Unit 2
UNIT 2 REACTOR TRIP DUE TO
STEAM FLOW/FEED FLOW MISMATCH

In accordance with 10 CFR 50.73, "Licensee Event Report System," the following report is submitted:

Licensee Event Report (LER) 316/2003-005-00: "Unit 2 Reactor Trip Due to Steam Flow/Feed Flow Mismatch."

Attachment 1 identifies the commitments included in this submittal.

Should you have any questions regarding this correspondence, please contact Mr. Toby K. Woods, Compliance Supervisor, at (269) 466-2798.

Sincerely,

Joseph N. Jensen
Site Vice President

RAM/jen

Attachments

IE22

c: J. L. Caldwell – NRC Region III
K. D. Curry – AEP Ft. Wayne
J. T. King - MPSC
MDEQ – WHMD/HWRPS
NRC Resident Inspector
Records Center - INPO
J. F. Stang, Jr. – NRC Washington DC

ATTACHMENT 1 TO AEP:NRC:2573-18

REGULATORY COMMITMENTS

The following table identifies those actions committed to by Indiana Michigan Power Company (I&M) in this document. Any other actions discussed in this submittal represent intended or planned actions by I&M. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Date
I&M will incorporate the enhanced standards for disconnecting and connecting leads into procedure 12-IHP-5021-IMP-001 (CRA 03365009-03).	March 19, 2004
I&M will incorporate the enhanced standards for disconnecting and connecting leads into procedure PMP-2291-PLN-001 (CRA 03365009-04).	March 19, 2004

NRC Form 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE

Unit 2 Reactor Trip Due To Steam Flow/Feed Flow Mismatch

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
12	30	2003	2003	-- 005 --	00	02	17	2004	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	1
10. POWER LEVEL	100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Toby Woods, Compliance Supervisor	(269) 466-2798

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If Yes, complete EXPECTED SUBMISSION DATE).	X	NO				

16. Abstract (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On December 30, 2003, at approximately 1328 Eastern Standard Time, the Unit 2 reactor automatically tripped on low steam generator water level coincident with feed flow less than steam flow for Steam Generator #22. Just prior to the trip, the Control Room Instrument Distribution (CRID) IV inverter transferred to the alternate power supply. A Feedwater Isolation Relay (K666X2), powered from CRID IV, opened. The opening of the relay initiated a seal-in closure signal to Steam Generator #22 and #23 Feedwater Isolation Valves (2-FMO-202 and 2-FMO-203), resulting in a loss of feedwater flow to Steam Generators #22 and #23 and produced a low steam generator water level trip coincident with feed flow less than steam flow. Following the reactor trip, the Unit Output Breakers failed to automatically open because a Main Turbine Stop Valve position indicator failed to provide a closed indication. The Unit Output Breakers were manually opened in accordance with the reactor trip response procedure. All safety components started and performed as expected. The initiator of the reactor trip was determined to be a momentary ground on the CRID IV bus, which occurred during the conduct of instrument calibration. The ground caused the automatic transfer of CRID IV to the alternate power source. The voltage drop on the CRID IV bus caused the Feedwater Isolation Relay to open. The root cause of the event has been determined to be inattention on the part of the Instrument and Control Technician who was performing the calibration. Corrective actions include the establishment of enhanced controls for the disconnecting and connecting of electrical leads.

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17. TEXT *(If more space is required, use additional copies of NRC Form (366A)*

Conditions Prior to Event

Unit 1 = 100% Power
Unit 2 – 100% Power

Description of Event

On December 30, 2003, at approximately 1328 EST, the Unit 2 reactor automatically tripped on low steam generator water level coincident with feed flow less than steam flow for Steam Generator #22.

Just prior to the trip, Instrument and Control Technicians were performing a calibration of 2-IFC-325 [BP - FCV], West Residual Heat Removal Pump (PP-35W) Discharge Flow Switch. While connecting a lead, a momentary ground occurred at the terminal box. The resulting ground fault current and subsequent inrush current caused the Control Room Instrument Distribution (CRID) IV [ED - INVT] inverter to transfer to the alternate power supply. The voltage transient on the CRID IV bus was enough to open the Feedwater Isolation Relay (K666X2). The opening of the relay initiated a seal-in closure signal to Steam Generator #22 and #23 Feedwater Isolation Valves (2-FMO-202 and 2-FMO-203) [ISV]. Control Room personnel observed the CRID IV abnormal annunciator and Steam Generator water level alarms and immediately implemented actions to respond to the alarms. An automatic reactor trip occurred due to a low steam generator level coincident with feed flow less than steam flow.

Following the reactor trip, the Unit 2 Output Breakers failed to automatically open due a Main Turbine Stop Valve [SB] position indicator failing to provide a closed indication. The Unit Output Breakers were manually opened in accordance with the reactor trip response procedure. All safety components started and performed as expected.

At 1501 hours on December 30, 2003, Indiana Michigan Power Company (I&M) made a four-hour notification to the NRC Operations Center, in accordance with 10 CFR 50.72(b)(2)(iv)(B) to report the actuation of the reactor protection system and resultant reactor scram.

Cause of Event

The cause of the Unit 2 automatic reactor trip was the loss of feedwater flow to Steam Generator #22. The cause of the loss of feedwater flow was the closure of the Feedwater Isolation Valve when the Feedwater Isolation Relay opened following a momentary electrical disturbance on CRID IV. The electrical disturbance was caused by a momentary ground fault created while connecting a lead during the calibration of 2-IFC-325, which is powered by CRID IV.

The root cause of the Unit 2 trip was inattention on the part of the Instrument & Control Technician performing the connection of live leads in the terminal box for 2-IFC-325. The technician brought the lead in close proximity to the terminal box edge and caused a ground fault. The current drawn as a result of the fault caused a voltage transient resulting in an unanticipated voltage drop below the design opening value of relay 2-K666-X2-B. The opening of this relay sent a close signal to the Feedwater Isolation Valves, which subsequently resulted in the reactor trip.

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17. TEXT *(If more space is required, use additional copies of NRC Form (366A)*

Analysis of Event

An assessment of this event was performed and it was determined that this event was bounded by the existing accident analysis associated with unplanned reactor trips with the main condenser available. The change in risk with respect to core damage and large early release frequency due to the inadvertent CRID inverter automatic transfer, together with the associated inadvertent closure of the Feedwater Isolation Valves and subsequent plant trip, have been qualitatively assessed and judged no different than any other unplanned reactor trips with the main condenser available. This assessment is based on the following considerations:

- The automatic plant trip, due to low main feedwater flow, functioned properly. The automatic trip features also functioned dependably, with the exception of a failed Main Turbine Stop Valve limit switch, which resulted in failure of the Unit Output Breaker to automatically open. However, the operators responded in an appropriate and timely manner, resulting in a safe and stable plant configuration.

- The inadvertent CRID inverter automatic transfer did not contribute to the increased likelihood of any initiators, other than transients that result in or from a reactor trip.

- Neither the CRID inverter automatic transfer nor the associated inadvertent closure of the Feedwater Isolation Valves degraded any system used to prevent core damage, assure containment integrity, or maintain defense-in-depth and safety margins.

Accordingly, I&M concludes that there was no impact on the health and safety of the public as a result of this event.

Corrective Actions

Plant Equipment:

- Verification of proper operation of the CRID.

- Replacement of Main Turbine Stop Valve position indicator and verification of its proper operation.

Immediate Actions:

- An event specific causal evaluation was performed.

- A human performance root cause was performed for this human error event as well as other recent similar human performance events within the Maintenance Department.

- As a result of the unacceptable human performance trend associated with maintenance work activities, a Stop Work Order was issued effective immediately to the Maintenance Department for all maintenance activities. This Stop Work Order is documented in Condition Report 04009044. The Stop Work Order will be rescinded when permanent preventive actions have been put into place to address barrier failures that occurred leading up to this event. The approval of the Plant Manager and the Site Vice President or Chief Nuclear Officer is required for rescinding this Stop Work Order.

- Enhanced standards were established for disconnecting and connecting leads.

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17. TEXT *(If more space is required, use additional copies of NRC Form (366A)*

Corrective Action to Prevent Recurrence:

- Incorporation of the enhanced standards for disconnecting and connecting leads into Procedure 12-IHP-5021-IMP-001, "Lead Lifting/Landing And Electrical Jumper/Fuse Installation And Removal." (Reference Corrective Action Item 3 of CR 03365009)
- Incorporation of the enhanced standards for disconnecting and connecting leads into Procedure PMP-2291-PLN-001, "Work Control Activity Planning Process." (Reference Corrective Action Item 4 of CR 03365009)

Previous Similar Events

None. A review of the LERs issued by I&M from 1998 to present, for both units, identified no similar occurrences (i.e., reactor trips due to maintenance personnel errors).